UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 21, 2014

Mr. Joseph E. Pacher
Vice President
R.E. Ginna Nuclear Power Plant, LLC
Exelon Generation Company, LLC
R.E. Ginna Nuclear Power Plant
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT – CORRECTION TO AMENDMENT NO. 116 ISSUED AUGUST 12, 2014, RE: REVISION TO TECHNICAL SPECIFICATION SECTION 3.6.5, "CONTAINMENT AIR TEMPERATURE." (TAC NO. MF0900)

Dear Mr. Pacher:

By letter dated August 12, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14191A682), the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 116 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. The amendment revised Technical Specification 3.6.5, "Containment Air Temperature," to increase the allowable containment air temperature from 120 °F to 125 °F, and was in response to your application dated February 28, 2013, as supplemented by letters dated June 19, and November 11, 2013, and January 22, March 14, March 26, and June 6, 2014.

The August 12, 2014, letter only provided a proprietary version of the license amendment and associated safety evaluation, and did not appropriately indicate the proprietary document markings.

Enclosure 1 to this letter provides a corrected proprietary version of the Amendment No. 116 in its entirety. Enclosure 2 to this letter provides a non-proprietary version of Amendment No. 116. Areas of change are indicated by revision bars in the right margin.

The corrections include changes to the August 12, 2014, transmittal letter, to include reference to the proprietary and non-proprietary versions, and corrects the safety evaluation so that the proprietary information is appropriately indicated by text enclosed within brackets.

NOTICE: Enclosure 1 to this letter contains Proprietary Information. Upon separation from Enclosure 1, this letter is DECONTROLLED.

This correction does not change the NRC staff's conclusions in the safety evaluation associated with the amendment. If you have any questions, please contact me at 301-415-1476.

Sincerely,

Mohan C. Thadani

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 116 (Proprietary)
2. Amendment No. 116 (Non-Proprietary)

cc w/Enclosure 2: Distribution via Listserv

**ENCLOSURE 2**

LICENSE AMENDMENT NO. 116,
DATED AUGUST 12, 2014
**(NON-PROPRIETARY)**

EXELON GENERATION COMPANY, LLC
R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

(ADAMS Accession No. ML14232A125)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 12, 2014

Mr. Joseph E. Pacher
Vice President
R.E. Ginna Nuclear Power Plant, LLC
Exelon Generation Company, LLC
R.E. Ginna Nuclear Power Plant
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT RE: REVISION TO TECHNICAL SPECIFICATION SECTION 3.6.5, "CONTAINMENT AIR TEMPERATURE." (TAC NO. MF0900)

Dear Mr. Pacher:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 116 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment is in response to your application dated February 28, 2013, as supplemented by letters dated June 19, and November 11, 2013, and January 22, March 14, March 26, and June 6, 2014.

The amendment revises Technical Specification 3.6.5, "Containment Air Temperature," to increase the allowable containment air temperature from 120 °F to 125 °F.

The NRC has determined that the related Safety Evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations,* Section 2.390, "Public inspections, exemptions, requests for withholding." Proprietary information is indicated by text enclosed within brackets. Accordingly, the NRC staff has also prepared a non-proprietary, publically available version of the SE, which is provided in Enclosure 3. The proprietary version of the SE is provided in Enclosure 2.

NOTICE: Enclosure 2 to this letter contains Proprietary Information. Upon separation from Enclosure 2, this letter is DECONTROLLED.

The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Mohan C. Thadani

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 116 to Renewed License No. DPR-18
2. Safety Evaluation (Proprietary)
3. Safety Evaluation (Non-Proprietary)

cc w/Encl 1 and 3: Distribution via Listserv

UNITED STATES
**NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

R.E. GINNA NUCLEAR POWER PLANT, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-244

R.E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 116
Renewed License No. DPR-18

1. The U.S. Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by Exelon Generation Company, LLC (Exelon, the licensee) dated February 28, 2013, as supplemented by letters dated June 19, and November 11, 2013, and January 22, March 14, March 26, and June 6, 2014 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-18 is hereby amended to read as follows:

Enclosure 1

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.116, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley

Benjamin G. Beasley, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: August 12, 2014

ATTACHMENT TO

LICENSE AMENDMENT NO. 116

RENEWED FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

| Remove | Insert |
|---|---|
| 3 | 3 |

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the areas of change.

| Remove | Insert |
|---|---|
| 3.6.5-1 | 3.6.5-1 |

(b) Exelon Generation pursuant to the Act and 10 CFR Part 70, to possess and use four (4) mixed oxide fuel assemblies in accordance with the RG&E's application dated December 14, 1979 (transmitted by letter dated December 20. 1979). as supplemented February 20,1980, and March 5, 1980;

(3) Exelon Generation pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Exelon Generation pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Exelon Generation pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Part 20. Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

(1) Maximum Power Level

Exelon Generation is authorized to operate the facility at steady-state power levels up to a maximum of 1775 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.116 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

(a) Exelon Generation shall implement and maintain in effect all fire protection features described in the licensee's submittals referenced in and as approved or modified by the NRC's Fire Protection Safety Evaluation (SE) dated February 14, 1979, and SE supplements dated December 17, 1980, February 6, 1981, June 22, 1981, February 27, 1985, and March 21, 1985 or

## 3.6 CONTAINMENT SYSTEMS

### 3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 125°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| CONDITION | | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|---|
| A. | Containment average air temperature not within limit. | A.1 | Restore containment average air temperature to within limit. | 24 hours |
| B. | Required Action and associated Completion Time not met. | B.1 | Be in MODE 3. | 6 hours |
| | | <u>AND</u> | | |
| | | B.2 | Be in MODE 5. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.5.1 | Verify containment average air temperature is within limit. | 12 hours |

# ENCLOSURE 3
# (NON-PROPRIETARY)

<u>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION</u>

<u>RELATED TO AMENDMENT NO. 116</u>

<u>TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18</u>

<u>R.E. GINNA NUCLEAR POWER PLANT, LLC</u>

<u>EXELON GENERATION COMPANY, LLC</u>

<u>R.E. GINNA NUCLEAR POWER PLANT</u>

<u>DOCKET NO. 50-244</u>

UNITED STATES
**NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 116

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18

R.E. GINNA NUCLEAR POWER PLANT, LLC

EXELON GENERATION COMPANY, LLC

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

## 1.0 INTRODUCTION

By application dated February 28, 2013 to the U.S. Nuclear Regulatory Commission (NRC) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13067A328), pursuant to Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), R.E. Ginna Nuclear Power Plant LLC, (Ginna LLC, the licensee) submitted a License Amendment Request (LAR) for a change to the R.E. Ginna Nuclear Power Plant (Ginna, the facility), Technical Specifications (TSs). The licensee proposed to revise the allowable containment average air temperature from "≤ 120 °F" to "≤ 125 °F" for TS 3.6.5 "Containment Air Temperature."

By supplemental letters dated June 19, and November 11, 2013, and January 22, March 14, March 26, and June 6, 2014 (ADAMS Accession Nos. ML13190A013, ML13323A083, ML14055A045, ML14079A522, ML14093A031, and ML14167A010, respectively), the licensee provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 26, 2013 (78 FR 70594).

## 2.0 REGULATORY EVALUATION

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products to the environment in the event of an accident. The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant accidents (LOCAs), or steam line accidents. The containment structure must continue to serve as a low leakage barrier against the release of fission products for as long as the postulated accident requires.

The NRC staff's review of the licensee's request covers the pressure and temperature conditions in the containment due to a spectrum of postulated LOCAs and secondary line breaks. The NRC's acceptance criteria for primary containment functional design are based on

the Appendix A to 10 CFR Part 50 general design criteria (GDC) 16 and 50 for the containment and its associated systems being able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. They are also based on GDC 38, which requires the provision of containment heat removal system(s) to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels. Specific review criteria are contained in Section 6.2.1.1.A, "PWR [Pressurized-Water Reactor] Dry Containments, Including Subatmospheric Containments" of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP).

The proposed increase in containment temperature by 5 °F would result in an increase in the Safety Injection accumulator temperature by 5 °F. An increase in the accumulator temperature has an effect on emergency core cooling system (ECCS) performance since maximum accumulator temperature is a parameter used as input in LOCA analyses. Acceptance criteria for ECCS for light water nuclear power reactors are contained in the NRC regulations in 10 CFR 50.46. Specifically, 10 CFR 50.46(a)(1)(i) states, in part, that:

> *ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.*

The regulation also requires that uncertainly is accounted for so that, when the calculated ECCS cooling performance is compared to the criteria in 10 CFR 50.46(b), there is a high level of probability that the limits set by the criteria would not be exceeded following a postulated LOCA. These criteria include requirements for peak cladding temperature (PCT), maximum cladding oxidation, core-wide oxidation (CWO), and long-term cooling.

Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity during Design Basis Accident Conditions," requires the licensees to determine if the containment air cooler cooling water systems are susceptible to water hammer or two-phase flow condition during postulated accident conditions, and if piping systems that penetrate containment are susceptible to thermal expansion of fluid so that over pressurization of piping could occur.

## 3.0 TECHNICAL EVALUATION

The design basis for the containment requires that the containment must withstand the pressures and temperatures of the design-basis accidents (DBAs) without exceeding the containment's design leak rate. The containment systems, and engineered safety features (ESF) are designed to ensure that the release of radioactive material subsequent to a DBA is prohibited from resulting in doses exceeding the guideline values of 10 CFR 50.67. The containment, ESF, and other associated safety systems are designed to ensure that the leakage of radioactive material to the environment is minimized. Consequently, there will be no undue risk to public health and safety.

3.1 Containment Design Basis Accident Analysis

Containment average air temperature is an initial condition used in the (DBA analysis to ensure that the total amount of energy within the containment is within the capacity of the Containment Spray (CS) and Containment Recirculation Fan Cooler (CRFC) systems. The containment average air temperature is also an important consideration in establishing the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA for the containment as described in the Ginna Updated Final Safety Analysis Report (UFSAR), Section 6.2.1.2.

This limiting condition for operation (LCO) ensures that the initial conditions assumed in the analysis of the containment response to a DBA are not violated during plant operations. The total amount of energy to be removed from the containment by the CS and CRFC systems during post- accident conditions is dependent upon the energy released to the containment due to an event, as well as the initial containment temperature and pressure. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Therefore, operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

Currently at Ginna, the maximum average temperature inside the containment is limited to 120 °F by operation of the CRFC cooling units. The maximum allowable Service Water (SW) suction temperature is 85 °F. During normal operation, the CRFC SW outlet valves, which are fully opened following a LOCA, are normally throttled to control containment temperature and pressure. Occasionally, during extended periods of high outside air temperature, all four coolers are used to limit the average containment temperature to 120 °F.

The licensee stated that historically, SW temperatures peak the first week of August; however, during the summer of 2012 the SW temperature peaked two weeks sooner due to an extended period of high temperatures. Correspondingly, the containment average air temperature was approaching the 120 °F limit. At this time, the Ultimate Heat Sink, which is Lake Ontario for Ginna, can experience a change in the depth of the different layers of water resulting in the water temperature significantly decreasing for a period of several days and the containment average air temperature may decrease. Based on the recent trend of increasing ambient and lake temperatures, there is a possibility that the containment average air temperature limit of 120 °F could be exceeded if there is an extended period of high temperatures without a corresponding change in the lake water temperature.

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and Main Steam Line Break (MSLB) which are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. The postulated DBAs are analyzed with regard to the capability of the ESF systems to mitigate the accident, assuming the worst case single active failure. Consequently, the ESF systems must continue to function within the environment resulting from the DBA. The environmental condition includes prevailing humidity, pressure, temperature, and level of radiation.

To support its proposed change, the licensee evaluated the accidents that are impacted by either the increase in initial containment air temperature or an increase in Safety Injection (SI) accumulator temperature. The SI accumulators are located in the Ginna containment, and are assumed to be at the same temperature as the containment. The analyses impacted by the

change in containment temperature that have been re-calculated include the LOCA and the MSLB to containment response analyses.

The licensee provided the results of its analysis in Table 1. This table compares the results obtained from the analyses performed in support of the licensee's extended power uprate (EPU) and the analyses performed in support of this proposed amendment. The licensee stated that, during the EPU licensing, the most limiting case was the double ended hot leg (DEHL) break for containment pressure, but with the proposed change in the containment initial air temperature and the changes to the mass and energy (M&E) release inputs (due to the correction of the errors in the EPITOME code), the double ended pump suction (DEPS) break is the most limiting for containment pressure. Both cases were analyzed with a loss of offsite power and a single failure of one of the two emergency diesel generators, denoted as minimum safeguards. Also, the DEPS break was the most limiting for containment temperature in the EPU analysis, and remains the most conservative. The licensee results show that the peak containment pressure and temperature remain below the containment design pressure of 60.0 pounds per square inch gauge (psig), and below the containment design and Equipment Qualification (EQ) maximum temperature of 286 °F.

Table 1, LOCA Analysis Results

| | Peak Pressure (psig) | Time (sec) | Peak Gas Temperature (°F) | Time (sec) | Pressure @ 24 hrs (psig) | Temp @ 24 hrs (°F) |
|---|---|---|---|---|---|---|
| DEHL (EPU) | 54.21 | 15.52 | 280.1 | 15.02 | N/A | N/A |
| DEHL (Proposed) | 54.25 | 16.01 | 280.4 | 16.01 | N/A | N/A |
| DEPS- Min Safeguards (EPU) | 53.88 | 1110 | 282.4 | 1110 | 7.77 | 159.4 |
| DEPS- Min Safeguards (Proposed) | 54.61 | 1220 | 283.6 | 1220 | 7.53 | 161.0 |

3.1.1 LOCA Mass and Energy Release and Containment Response

The licensee evaluated the design-basis LOCA relative to the containment peak pressure and temperature response with the increased initial containment air temperature condition.

The containment analysis consists of two parts. First the M&E release from a high-energy line break (HELB) is calculated. Second, the containment conditions resulting from this release of M&E into the containment are calculated.

The licensee used the GOTHIC (Generation of Thermal Hydraulic Information for Containments) 7.2a computer code to calculate the pressure and temperature conditions within the containment resulting from a postulated HELB. GOTHIC is a general purpose computer program for the prediction of the thermal hydraulic conditions in nuclear power plant containments. GOTHIC solves the conservation equations for mass, momentum and energy for multi-component, multi-phase flow. It is subject to 10 CFR Part 50, Appendix B and 10 CFR Part 21 requirements.

In its LAR, the licensee stated that the increased containment temperature has no impact on the design basis short-term LOCA M&E releases that would be used for short-term sub-compartment analyses. Likewise, the increased accumulator temperature has no impact on the design basis short-term LOCA M&E releases that would be used for short-term sub compartment analyses because the releases are calculated for only one second to three seconds and accumulators would not begin to inject until many seconds into the transient.

The licensee stated that the long term LOCA M&E and containment response was analyzed in two stages. The first of which is from 0 to 3600 seconds, i.e. the time at which energy in the primary heat structures and steam generator secondary system is released / depressurized to atmospheric pressure, (i.e. 14.7 pounds per square inch absolute (psia) and 212 °F). This portion of the long-term LOCA M&E release was analyzed using the WCAP-10325-P-A methodology. The second stage, i.e. post 3600 seconds, was analyzed using GOTHIC version 7.2a.

This methodology for the long-term LOCA M&E release analysis was approved for use at Ginna in the EPU Safety Evaluation Report (ADAMS Accession No. ML061380249).

The licensee stated that the error identified by Westinghouse in the EPITOME computer code in the WCAP-10325-P-A methodology, which under-predicted the long-term LOCA M&E release has been corrected in the proposed analysis.

In addition to the change in initial containment air temperature in the model, the licensee stated that in order to increase the margin between the calculated peak pressure and the containment design pressure limit, the stored energy in the secondary side metal that is above the secondary side liquid level is assumed to be released to the containment atmosphere over 24 hours. This energy was previously released during the first 3600 seconds. The licensee stated that the additional change is not a departure from the method described in WCAP-10325-P-A, because the secondary inventory is still assumed to release all available energy (i.e. down to 212 °F and 14.7 psia) within the first 3600 seconds, and all steam generator (SG) energy is available for release.

On March 24, 2014, the NRC staff and the licensee discussed, by phone call, the initial heat sinks temperature used in the analyses, and the licensee verified that initial heat sinks temperature was increased to 125 °F and that the peak containment temperature limit would not exceed the containment wall temperature design limit of 286 °F.

The NRC staff has reviewed the proposed changes from the current licensing basis (CLB) identified in the LAR and finds that the licensee used the overall approved methodology in WCAP-10325-P-A, with revised input assumptions reflecting modeling refinements based on better and more detailed data, and correcting an error in the EPITOME code for LOCA M&E releases. Therefore, the NRC staff concludes that these changes are acceptable.

The licensee stated that with the new model, and changes to the LOCA M&E release inputs, the DEPS break is the most limiting for containment pressure. Also, the DEPS break was the most limiting for containment temperature in the EPU analysis, and remains the most conservative. The peak containment pressure and gas temperature, 54.61 psig and 283.6 °F, remain below the containment design pressure of 60.0 psig, and below the containment design and EQ maximum temperature of 286 °F. This is conservative and therefore acceptable.

The containment pressure using minimum safeguards at 24 hours following the start of the LOCA is 7.53 psig. This is significantly less than half of the peak containment pressure. It is therefore acceptable to follow the guidance of SRP Section 6.2.1.1.A to assume that the containment leakage rate after 24 hours is less than half the TS containment leakage rate limit of $L_a$.

The NRC staff has reviewed the licensee's evaluation and finds it acceptable because the licensee used the NRC-approved methodology, and showed acceptable results.

3.1.2 MSLB Mass and Energy Release and Containment Response

The licensee performed analyses of the containment response to the MSLB analysis in support of an increased initial containment air temperature condition using NRC-approved methods for the MSLB M&E release into the containment. The containment pressure response was analyzed using the GOTHIC 7.2a computer code, in accordance with methods previously approved by the NRC as part of the Ginna EPU (ADAMS Accession No. ML061380249).

In the LAR, the licensee described the Ginna GOTHIC containment model and the new assumptions used in the analyses. These included the assumed initial containment air temperature condition and parameters describing the CS systems, which consist of the CS flow rate reduced from a constant 1300 gpm to a spray flow as a function of containment pressure at reduced pump performance.

The licensee stated that the dose analyses and post-LOCA long-term cooling analyses will continue to use 1300 gpm. The reduction in spray flow rate in the MSLB analysis is to provide for a potential future margin increase in the CS pumps. In addition, the initial heat sinks (e.g. containment wall) temperature was increased in the analyses to 125 °F, and the analyses results show that even with the increase in the containment wall temperature, it would not exceed the 286 °F design bases limit.

To support the increase in the initial containment temperature to 125 °F, the licensee only re-analyzed the most limiting case determined in the current analysis (documented in UFSAR Section 6.2.1.2.3) because the change in the revised inputs, i.e., containment air temperature and initial heat sinks temperature has the same impact on all cases. This most limiting case considers a break area of 1.4 $ft^2$ at 70-percent power, assuming a loss of one train of containment fan coolers and one CS pump.

To gain additional margin for containment peak pressure in the limiting case, the licensee adjusted two additional inputs: (1a) it assumed a delay in the turbine-driven auxiliary feedwater pump start to coincide with the faulted SG reaching the low-low level setpoint which reduces the mass release from the faulted SG: and (2b) it assumed a change in the SI enthalpy, reducing it from 77.9 Btu/lbm to 75.9 Btu/lbm. The NRC staff has previously approved these input adjustments (ADAMS Accession No. ML061380249).

The results for the new MSLB containment pressure response case yielded a peak containment pressure of 59.68 psig, just below the containment design pressure of 60.0 psig.

The NRC staff's review of the licensee's MSLB analysis, for containment response, determined that the analyses were performed using acceptable analytical models, and the acceptance

criteria. Therefore, the NRC staff concludes that containment pressure safety limits have been met.

3.1.3 Post-LOCA Long-Term Cooling

Fan cooler systems, spray systems, and residual heat removal (RHR) systems are provided to remove heat from the containment atmosphere and from the water in the containment sump. The NRC staff's review in this area focused on: (1) the effects of the proposed increased initial containment air temperature on the analyses of the available net positive suction head (NPSH) to the containment heat removal system pumps, and (2) the analyses of the heat removal capabilities of the spray water system and the fan cooler heat exchangers.

In a request for additional information, the NRC staff requested that the licensee discuss the impact of the increased initial containment temperature on the sump water temperature and on the available NPSH for the RHR and CS pumps during the sump recirculation phase of post-LOCA cooling. In its response (ADAMS Accession No. ML14079A522, Response # 2), the licensee stated that the NPSH margin for the CS and RHR pumps was re-evaluated because the sump water temperature increased slightly over its current value due to the higher initial containment air temperature.

The RHR pumps take suction from the containment sump and provide water to the suction of the high-head SI pumps. The licensee stated that for the most limiting conditions for the RHR pump NPSH (with and without SI) required during recirculation, the containment pressure is conservatively taken as zero psig, and a low reactor coolant system (RCS) pressure conservatively maximizes the RHR pump flow. Due to the increase in sump water temperature that resulted from the increase in initial containment air and accumulator temperature the RHR pumps maintain an NPSH margin of greater than 3.50 feet. The NRC staff concludes that the licensee's analysis is acceptable because the licensee demonstrated adequate NPSH margin for the RHR pumps while not crediting containment accident pressure.

The licensee stated that the containment cooling does not credit CS pump operation during sump recirculation following a DBA, therefore the sump water temperature does not impact the NPSH margin for the CS pumps.

Based on the above discussion, the NRC staff concludes that the containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained and will continue to meet the requirements of GDCs 16, 38, and 50.

3.2 Evaluation of the Effect of the TS Change And Thermal Conductivity Degradation

The NRC staff reviewed the effect of the TS change on Large Break Loss of Coolant Accident (LBLOCA) Core Response Analysis, Small Break Loss of Coolant Accident (SBLOCA) Core Response Analysis, and Post-LOCA Long-Term Cooling. The results of the review are presented below.

The technical evaluation is conducted, as applicable, for the effect of thermal conductivity degradation (TCD), followed by an evaluation for the effect of increasing the containment air temperature from 120 °F to 125 °F. Either effect can cause an increase in PCT.

Irradiation damage and the progressive buildup of fission products within fuel pellets result in reduced thermal conductivity of the pellets. Legacy fuel rod thermal-mechanical codes do not include this reduction in thermal conductivity with increasing exposure because earlier test data were inconclusive as to the significance of the effect. Beginning in the 1990s, measurements collected from instrumented fuel assemblies indicated steady degradation in the thermal conductivity of uranium fuel pellets with increasing exposure. This phenomenon is known as TCD and is discussed further in NRC Information Notice (IN) 2009-23, "Nuclear Fuel Thermal Conductivity Degradation." Its potential effects in realistic ECCS evaluation models are described in IN 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting From Nuclear Fuel Thermal Conductivity Degradation."

In LBLOCA analyses, the dominant phenomena affecting PCT during the blowdown phase is stored energy in the fuel (NUREG/CR-5249, December 1989). Stored energy in the fuel is also the phenomena most affected by TCD. This causes TCD to be included in evaluation models used for LBLOCA analyses. Alternatively, the more dominant source of energy in SBLOCA is decay heat. This causes TCD to not be significant in SBLOCA analyses.

3.2.1 LBLOCA Core Response Analysis

An LBLOCA is defined as a breach in the reactor coolant pressure boundary with a total cross-sectional area greater than 1.0 $ft^2$. Evaluation of the licensee's LBLOCA analyses consists of three parts: (1) EPU, (2) additional changes and/or errors to the ECCS evaluation model including TCD, and (3) increase in SI accumulator temperature. Each part is described in detail below.

The licensee uses the NRC-approved Automated Statistical Treatment of Uncertainty Method (ASTRUM), documented in WCAP-16009-NP-A (ADAMS Accession No. ML050910157), to evaluate ECCS performance. ASTRUM relies on an approach based on order statistics, in which a set number of cases with randomly varied initial conditions are analyzed using the WCOBRA/TRAC (WC/T) reactor system analysis code. The number of cases is chosen so that the highest predicted PCT within the case set becomes a predictor of the 95/95 upper tolerance limit for the PCT associated with a hypothetical population of LOCA scenarios. The result is used to show compliance with the 10 CFR 50.46(b)(1) acceptance criterion concerning PCT.

The licensee first completed an LBLOCA analysis using ASTRUM to support an EPU request in 2007. The licensee's LBLOCA analysis implementing ASTRUM was approved by the NRC staff under License Amendment No. 96 dated May 31, 2006 (ADAMS Accession No. ML061180353). This analysis included running **[[** **]]** in accordance with the ASTRUM evaluation methodology. The NRC staff's safety evaluation (SE) and acceptance (ADAMS Accession No ML061380249) establishes the analysis of record (AOR) PCT of 1870 °F for the LBLOCA under EPU conditions.

Since approval of the EPU, additional changes and/or errors to the ECCS evaluation model have been reported by the licensee through the regulations of 10 CFR 50.46. An August 16, 2012 report submitted by the licensee pursuant to 10 CFR 50.46 shows changes to the ECCS evaluation model pertaining to HOTSPOT fuel relocation, TCD, peaking factor burndown, and design input changes (ADAMS Accession No ML12233A621). TCD was accounted for exclusively in which **[[**

**]]** including the additional changes and/or errors to the ECCS evaluation model resulting in an LBLOCA PCT of 2041 °F.

For the purpose of this LAR, when the containment air temperature increases, the SI accumulator temperature also increases since the SI accumulator is located inside the containment.

The maximum accumulator temperature is a parameter used as input in LBLOCA analysis. If the accumulator temperature is higher during an LBLOCA, fuel temperatures will increase for a longer period of time. This will ultimately cause PCT to be higher.

For the third part of the LBLOCA evaluation, the licensee increased the maximum accumulator temperature from 120 °F to 125 °F. This is done by changing an input in the ASTRUM evaluation model. From the **[[ ]]** re-executed for the TCD evaluation, the licensee chose **[[ ]]** to be re-executed to account for the increase in accumulator temperature. The **[[ ]]** were chosen based on the highest PCT cases from the TCD subset (ADAMS Accession No. ML14093A031). The results of these analyses showed a maximum PCT of 2116 °F. This is 75 °F higher than the PCT reported previously (ADAMS Accession No. ML12233A621).

**[[**

**]]**

Of the original **[[ ]]** evaluated by the licensee in the AOR, the highest accumulator temperature was not a consideration when choosing the subset of cases to be re-executed for TCD. The TCD cases to be re-executed were chosen based on the highest PCT cases. **[[**

**]]** The results of the analyses of these additional cases show a maximum PCT of 1547 °F. This provides assurance that the most severe postulated LOCA has been calculated and that the results show that there is a high probability that the calculated PCT shall not exceed 2200 °F.

**[[**

**]]**

The 5 °F increase in accumulator temperature resulted in an ultimate increase in PCT of 75 °F. This results in a PCT of 2116 °F following a postulated LBLOCA event, which is below the 10 CFR 50.46(b)(1) limit of 2200 °F.

A summary of results for the three parts of the LBLOCA analyses described above is provided in the table below. Based on the above discussion, the NRC staff finds these results acceptable in regards to the ECCS.

| 10 CFR 50.46 Requirement | EPU | Additional Changes Including TCD | Increase in Accumulator Temp |
|---|---|---|---|
| PCT (°F) | 1870 | 2041 | 2116 |
| MLO (%) | 2.89 | - | 7.38 |
| CWO (%) | 0.30 | - | 0.97 |

3.2.2 SBLOCA Core Response Analysis

An SBLOCA is defined as a breach in the reactor coolant pressure boundary with a total cross-sectional area less than 1.0 $ft^2$. As described in the previous section, when the containment air temperature increases, the SI accumulator temperature also increases. An evaluation was performed with an increase in accumulator temperature from 120 °F to 125 °F for SBLOCA.

Note that TCD is not considered in SBLOCA analyses because the dominant source of energy in SBLOCA is decay heat, which is not affected by TCD.

The results of the licensee's SBLOCA analyses are included in the SE for the licensee's EPU (ADAMS Accession No. ML061380249). The limiting break size is a 2-inch break resulting in a PCT of 1167 °F. For break sizes of 2-inches or smaller, accumulator injection does not occur until 1182 seconds after the PCT occurs (UFSAR, Revision 23 ), therefore, the change in accumulator water temperature does not impact the PCT for these break sizes.

For break sizes 4-inches and larger, accumulator injection is required to prevent the core from becoming uncovered. In these cases, the accumulator flow rate is the dominating factor in the analyses. The amount of water entering the RCS to fill the downcomer and core and mixing with the cold leg fluid that is much hotter outweighs the 5 °F increase in accumulator temperature. This leaves the 2-inch break size as the limiting case.

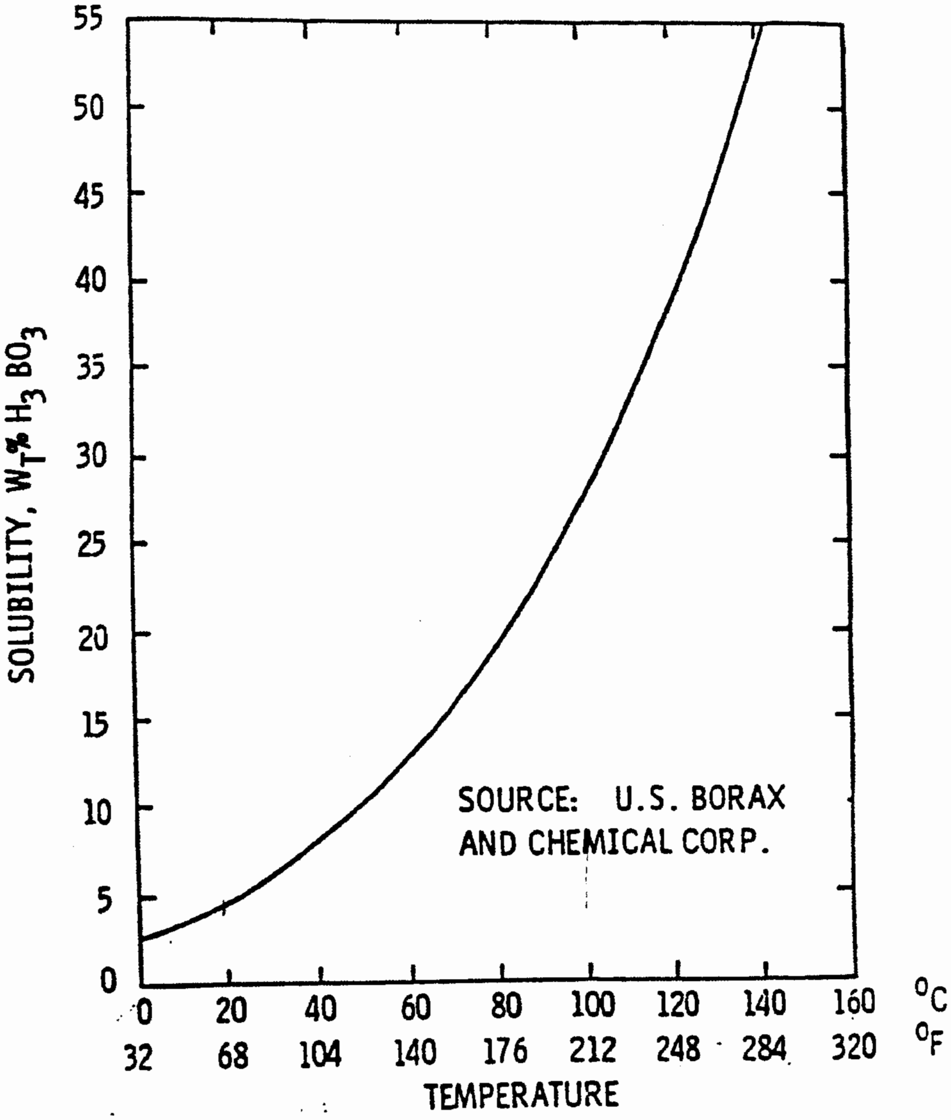
For the 3-inch break size where accumulator injection occurs only 75 seconds before PCT, core recovery depends on the timing of accumulator injection. If the accumulator temperature increases, fuel temperatures increase for a longer period of time once SI injection occurs. This will ultimately cause PCT to increase. The PCT for a break size of 3 inches is 1117 °F. There is a 50 °F margin between this case and the limiting PCT of 1167 °F for the 2-inch break. The result of a 5 °F increase in accumulator temperature will have a negligible impact on PCT when compared to the 50 °F margin to the limiting PCT for this case. This is because, similar to break sizes of 4-inches and larger, the amount of mass entering the system and mixing with the cold leg fluid that is much hotter outweighs a 5 °F increase in accumulator temperature.

The licensee concludes that the PCT for an SBLOCA remains 1167 °F for a break size of 2 inches. Based on the above discussion, the NRC staff finds the results of the license's analysis acceptable.

3.2.3 Post-LOCA Long-Term Cooling

Post-LOCA subcriticality and post-LOCA boric acid precipitation are evaluated for post-LOCA long-term cooling. Boric acid precipitation is a concern because when debris mixes with boric acid at certain temperatures, chemical composition causes blockage of the sump pumps. An increase in temperature results in an increase in solubility, as seen in the below figure. When the boron concentration remains constant, an increase in accumulator temperature will increase the solubility of the solution and the ability of chemical composition to form decreases. The NRC staff finds that an increase in accumulator temperature has a beneficial effect on the solubility of the boric-acid solution since solubility increases when temperature increases.

Additionally, the licensee performed post-LOCA subcriticality analyses. The results show a negligible change in the sump mixed-mean boron concentration curve. The results of this evaluation are reviewed by the licensee each cycle to ensure that the post-LOCA core will remain subcritical upon entering the sump recirculation phase of ECCS injection. Therefore, the NRC staff finds this analysis acceptable.



3.2.4 Conclusion

The NRC staff reviewed the licensee's assessment of the effect that the proposed increase in maximum allowable containment air temperature would have on ECCS performance, as described in the preceding paragraphs. Based on its review, the NRC staff determined that the licensee has acceptably evaluated the effect that the proposed change would have on the plant's compliance with the acceptance criteria contained in 10 CFR 50.46(b). The NRC staff concludes, therefore, that the proposed change is acceptable with respect to ECCS performance.

However, it must be noted that the licensee did not use an acceptable evaluation model as required by 10 CFR 50.46(a)(1)(i). The acceptable evaluation model for LBLOCA remains ASTRUM as implemented in accordance with License Amendment No. 96 dated May 31, 2006 (ADAMS Accession No. ML061180353). The licensee has committed to re-analyze ECCS performance, pursuant to 10 CFR 50.46(a)(3)(ii), in a letter dated June 19, 2013 (ADAMS Accession No. ML13190A013) to address the effects of any significant changes or errors previously reported. The commitment states that:

> *R.E. Ginna Nuclear Power Plant, LLC will perform a LBLOCA re-analysis that applies NRC approved evaluation methodology, which includes the effects of nuclear fuel thermal conductivity degradation (TCD), within 24 months of the completion of the following three milestones:*
>
> 1) *Submittal by Westinghouse to the NRC for review and approval, of revised fuel performance analysis and LBLOCA evaluation model methodologies that include the effects of TCD.*
> 2) *NRC approval of the revised Westinghouse fuel performance analysis methodology that includes the effects of TCD. The new methodology would replace the current licensing basis methodology described in WCAP-16009-P-A, which is used to develop inputs to the LBLOCA evaluation model.*
> 3) *NRC approval of the revised Westinghouse LBLOCA evaluation model methodology that includes the effects of TCD. The new methodology would replace the current licensing basis methodology described in WCAP-16009-P-A.*

The NRC staff's acceptance of the present evaluation was not affected by this commitment to re-analyze ECCS performance.

3.3 GL 96-06, "Assurance of Equipment Operability and Containment Integrity during Design - Basis Accident Conditions"

The licensee's technical evaluation did not include a discussion of the effect of raising the initial containment average air temperature [TS 3.6.5 LCO limit] on the GL 96-06 requirements. By letter dated September 24, 2013, the NRC staff asked the licensee to discuss and justify the proposed increase in initial containment average air temperature [TS 3.6.5 LCO limit], considering GL 96-06 and the extended power uprate power level. In its response dated November 11, 2013, the licensee stated that prior evaluations for GL 96-06 used a peak containment temperature of 286 °F when predicting the issues associated with GL 96-06, namely, (1) the potential for reduced heat transfer due to two phase conditions in the

Containment Recirculation Fan Coolers, (2) the potential effects of water hammer loads, and (3) the potential for thermal over pressurization. The licensee further stated that prior GL 96-06 evaluations were not dependent on initial containment temperature other than its effect upon peak containment temperature. Since new DBA analyses predict peak containment temperature to remain below 286 °F, the licensee concluded that the prior GL 96-06 analysis bounds the proposed increase in allowable containment average air temperature [TS 3.6.5 LCO limit]. The NRC staff finds this analysis acceptable.

Based on the above evaluation, the NRC staff concludes that at the revised containment air temperature of 125 °F the containment structure will continue to be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated LOCAs, or steam line accidents. The containment structure will continue to serve as a low leakage barrier against the release of fission products for as long as the postulated accident requires.

### 3.4 Conclusion

The NRC staff's review covered the pressure and temperature conditions in the containment due to a spectrum of postulated LOCAs and secondary line breaks. The NRC staff concludes that, with the proposed revision, the acceptance criteria will continue to be met (1) for primary containment functional design based on GDC 16 and 50 for the containment and its associated systems being able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA; and (2) for GDC 38 for the containment heat removal system(s) to rapidly reduce the containment pressure and temperature to within their safe limits. Additionally, the licensee's evaluation of GL 96-06 demonstrates that the requirement of assurance of equipment operability and containment integrity during DBA conditions will be met, as required by GL 96-06.

Accordingly, the NRC staff concludes that the proposed changes to the TS requested by the licensee are acceptable.

## 4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

## 5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (78 FR 70594, dated November 26, 2013). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Guzzetta
R. Torres

Date: August 12, 2014

The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

**/RA/**

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 116 to Renewed License No. DPR-18
2. Safety Evaluation (Proprietary)
3. Safety Evaluation (Non-Proprietary)

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Package No.: ML
**Amendment NoS.: ML14191A682 (Proprietary); ML14232A097 (Corrected Proprietary); ML14232A125 (Non-Proprietary);**
**Tech Spec No.: ML052720231**

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| DATE | 07/21/2014 | 08/11/2014 | 04/15/14 | 06/20/14 | 12/11/13 |
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| NAME | JWachutka | BBeasley | MThadani | | |
| DATE | 07/25/2014 | 8/12/14 | 8/12/14 | | |

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This correction does not change the NRC staff's conclusions in the safety evaluation associated with the amendment. If you have any questions, please contact me at 301-415-1476.

Sincerely,

*/RA/*

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
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2. Amendment No. 116 (Non-Proprietary)

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ADAMS Accession Nos. Correction letter ML14232A229 (Proprietary)
Correction letter ML14232A331 (Non-proprietary)
Revised Proprietary Amendment 116 ML14232A097
Non-proprietary - Amendment No. 116 ML14232A125

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| DATE | 8/20/14 | 8/20/14 | 8/21/14 | 8/21/14 |

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